U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

5.2.2 OVERPRESSURE PROTECTION

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (SRXB)

Secondary - None

I. AREAS OF REVIEW

A. Overpressure protection for the reactor coolant pressure boundary (RCPB), during power operation of the reactor, is ensured by application of relief and safety valves and the reactor protection system. For boiling water reactors (BWRs), the area of review includes relief and safety valves on the main steam lines and piping from these valves to the suppression pool. For pressurized water reactors (PWRs), the area of review includes pressurizer relief and safety valves and the piping from these valves to the quench tank~~,~~ on the primary,[1] and steam generator relief and safety valves on the secondary.

The adequacy of the proposed preoperational and initial startup test programs is examined as a part of this review. The reviewer also evaluates the proposed technical specifications to assure that they are adequate with regard to limiting conditions of operation and periodic surveillance testing.

B. Overpressure protection for the RCPB, during low temperature operation of the plant (startup, shutdown), is ensured by the application of pressure relieving systems that function during the low temperature operation. For PWRs the area of review includes relief valves with piping to the quench tank, the makeup and letdown system, and the RHR system which may be operating when the primary system is water solid. For

BWRs, no special area of review is required since BWRs never operate in water-solid conditions.

Review Interfaces:[2]

SRXB also performs the following review under the SRP section indicated:

1. Reviews the reactor coolant depressurization system for PWRs as part of its primary review responsibility for SRP Section 6.8 (proposed).[3]

2. The SRXB also reviews the design of systems that interface with the reactor coolant system with regard to the capability of the interfacing system to withstand full RCS pressure as part of its primary review responsibility for SRP Section 3.12 (proposed).[4]

In addition, the SRXB will coordinate its review with the evaluations of other branches that have primary review responsibility for other portions of the overpressure protection as follows:

1. The ~~Human Factors Assessment Branch (LHFB)~~Quality Assurance and Maintenance Branch (HQMB)[5], as part of its primary review responsibility for SRP Section 14.2, reviews proposed preoperational and initial startup test programs to assure that overpressure components will perform their safety function.

2. The Mechanical Engineering Branch (EMEB), as part of its primary review responsibility for SRP Sections 3.2.1 and 3.2.2, reviews seismic design criteria for components of the overpressure protection system.

3. The EMEB, as part of its primary review responsibility for SRP Section 3.10~~ and the Plant Systems Branch (SPLB) for SRP Section 3.11~~[6], reviews[7] ~~installation criteria~~seismic and dynamic qualification[8] for components of the overpressure protection system.

4. The Plant Systems Branch (SPLB), as part of its primary review responsibility for SRP Section 3.11, reviews environmental qualification for components of the overpressure protection system.[9]

5. The Materials and Chemical Engineering Branch (EMCB) reviews the fracture toughness of the RCPB and reactor vessel as part of its primary review responsibility for SRP Sections 5.2.3 and 5.3.1. The EMCB also reviews the pressure-temperature limits and pressurized thermal shock analysis as part of its primary review responsibility for SRP Section 5.3.2.[10]

6. The Instrumentation ~~and~~& Controls~~ Systems~~ Branch (~~S~~HICB)[11], as part of its primary review responsibility for SRP Section 7.6, reviews the adequacy of controls and instrumentation for the automatic and manual actuation of overpressure protection components.

7. The Technical Specifications Branch (~~O~~TSB),[12] as part of its primary review responsibility for SRP Section 16.0, reviews technical specifications.

8. The ~~Performance and Quality Evaluation Branch~~[13] ~~(PQEB)~~HQMB[14], as part of its primary review responsibility for SRP Sections 17.1 and 17.2, reviews quality assurance requirements.

For those areas of review identified above as being reviewed as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding primary branch.

II. ACCEPTANCE CRITERIA

The SRXB acceptance criteria for the overpressure protection system are based on meeting the relevant requirements of the following regulations:

1. General Design Criterion 15, as it relates to the reactor coolant system and associated auxiliary, control, and protection systems being designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

2. General Design Criterion 31, as it relates to the reactor coolant pressure boundary being designed with sufficient margin to assure that ~~boundary~~it behaves in a nonbrittle manner and that[15] the probability of rapidly propagating fracture is minimized.

~~Applications for construction permit should meet recommendations of Task Action Plan items II.D.1 and II.D.3 of NUREG-0718 (Ref. 4). Applications for operating license shall meet recommendations of Task Action Plan items II.D.1 and II.D.3 of NUREG-0737 (Ref. 5).~~

Applicants should meet the recommendations of the following TMI action plan items of NUREG 0737 (Reference 6):

1. II.D.1 regarding testing to qualify reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents. 10 CFR 50.34(f)(2)(x) establishes an equivalent requirement for those applicants subject to the requirements of 10 CFR 50.34(f).[16]

2. II.D.3 regarding the provision of direct indication of relief and safety valve position. 10 CFR 50.34(f)(2)(xi) establishes an equivalent requirement for those applicants subject to the requirements of 10 CFR 50.34(f).[17]

Other specific acceptance criteria necessary to meet the requirements of GDC 15 and 31 are as follows:

A. For overpressure protection, during power operation of the reactor, the relief valves shall be designed with sufficient capacity to preclude actuation of safety valves, during normal operational transients, when assuming the following conditions at the plant:

   a. The reactor is operating at licensed core thermal power level.

b. All system and core parameters are at values within normal operating range that produce the highest anticipated pressure.

c. All components, instrumentation, and controls function normally.

Safety valves shall be designed with sufficient capacity to limit the pressure to less than 110% of the RCPB design pressure (as specified by the ASME Boiler and Pressure Vessel Code [Reference~~. 2~~15][18]), during the most severe abnormal operational transient with reactor scram. Also, sufficient margin shall be available to account for uncertainties in the design and operation of the plant assuming:

(1) The reactor is operating at a power level that will produce the most severe overpressurization transient.

(2) All system and core parameters are at values within normal operating range, including uncertainties and technical specification limits that produce the highest anticipated pressure.

(3) The reactor scram is initiated by the second safety-grade signal from the reactor protection system.

(4) The discharge flow is based on the rated capacities specified in the ASME Boiler and Pressure Vessel Code, for each type of valve.

~~3.~~[19]Full credit is allowed for spring-loaded safety valves designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code (Reference~~. 7~~16)[20].

B. The low temperature, overpressure protection (LTOP) system shall be designed in accordance with the requirements of Branch Technical Position RSB 5-2 attached to this SRP section (Reference~~. 3~~5)[21]. The LTOP system shall be operable during startup and shutdown conditions below the enable temperature defined in paragraph B.2. of RSB 5-2.

<u>Technical Rationale:</u>[22]

The technical rationale for application of the above acceptance criteria to the overpressure protection system is discussed in the following paragraphs:

1. GDC 15 requires that the reactor coolant pressure boundary be designed, constructed, and tested with sufficient margin to assure that design conditions are not exceeded during normal operation or anticipated operational occurrences. The overpressure protection system is relied upon to maintain reactor coolant system pressure within acceptable design limits during certain analyzed transients. Application of GDC 15 to the overpressure protection system provides assurance that the reactor coolant pressure boundary will have an extremely low probability of failure during transients.

2. GDC 31 requires that the reactor coolant pressure boundary be designed with sufficient margin to preclude brittle fracture during expected operational, maintenance, testing, and

accident conditions. During certain conditions in which the reactor coolant pressure boundary might behave in a brittle manner, the overpressure protection system is relied upon to maintain the reactor coolant system pressure below brittle fracture limits. Meeting GDC 31 will ensure that the reactor coolant pressure boundary is not subject to failure by brittle fracture.

III. REVIEW PROCEDURES

The procedures below are used during the construction permit (CP) review to assure that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report meet the acceptance criteria given in subsection II of this SRP section.

For operating license (OL) applications, the procedures are used to verify that the initial design criteria and bases have been appropriately implemented in the final design as set forth in the final safety analysis report and in the report on overpressure protection. The latter report is required by the ASME Code and is used as the basis for many of the individual review steps outlined below during the OL review.

The OL review also includes the proposed technical specifications, to assure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing.

The following steps are taken by the SRXB reviewer in determining that the acceptance criteria of subsection II have been met. These steps should be applied to CP and OL reviews as appropriate. Previously reviewed designs may be used as a guide; however, the reviewer must verify that any changes are justified.

1. The piping and instrumentation diagrams are examined to determine the number, type, and location of the safety and relief valves in both the primary and secondary systems, and of discharge lines, instrumentation, and other components.

2. All other functions of the components, instruments, or controls used for overpressure protection and the interfaces with all other systems are identified. The effects of these other functions or systems on operation of the overpressure protection system are determined. For PWRs, failure of the makeup and letdown system or the RHR system is examined to assure overpressure protection during low temperature operation of the plant.

3. The capacities, setpoints, and setpoint tolerances for all safety and relief valves are identified.

4. All of the reactor trip signals which occur during overpressure transients, including their setpoints and setpoint tolerances, are identified.

5. All transients analyzed in Chapter 15 of the SAR that result in an increase in the pressure experienced by the RCPB are examined. The predicted peak pressures are identified and the operating conditions and setpoints used in the analysis are reviewed to assure that they are suitably conservative.

6. The proposed plant technical specifications are reviewed to:
   a. Confirm the suitability of the limiting conditions of operation, including the proposed time limits and reactor operating restrictions for periods when system equipment is inoperable due to repairs and maintenance.

   b. Verify that the frequency and scope of periodic surveillance testing is adequate.

   c. Verify compliance with the technical specification guidance of Generic Letter 90-06 (Reference 14) for PORVs, PORV block valves, and the LTOP system (PWRs only).[23]

   d. Verify compliance with TMI action plan item II.K.3.3 of NUREG 0737 regarding reporting of safety relief valve challenges and failures. Generic Letters 82-16 and 83-02 (References 12 and 13) provide descriptions of this NUREG 0737 item and includes guidance regarding appropriate Technical Specifications to address the reporting requirements of II.K.3.3. Section 5.9.1 of Standard Technical Specifications (References 7 to 11), regarding Monthly Operating Reports, provides related guidance on an appropriate technical specification to address this issue for those applicants implementing Improved Technical Specifications.[24]

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP Section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.[25]

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and the review supports the following kinds of statements and conclusions, which should be included in the staff's safety evaluation report:

The staff concludes that the overpressurization protection system is acceptable and meets the relevant requirements of GDC 15 and 31 and Appendix G to 10 CFR Part 50. This conclusion is based on the following:

1. BWRs

   The overpressure protection system prevents overpressurization of the reactor coolant pressure boundary under the most severe transients and limits the reactor pressure during normal operational transients. Overpressure protection is provided by ______ safety and relief valves located on the four main steam lines between the reactor vessel and the first isolation valve inside the drywell. The relief and safety valves are distributed among the four main steam lines such that a single accident cannot disable the automatic overpressure protection function.

The valves discharge through piping to the suppression pool. The valves have setpoints that range from ______ to ______ kPag (______ to ______ psig)[26]. Their total capacity at their setpoint is ______ % of rated steam flow.

To determine the ability of the overpressure protection system to prevent overpressurization, the applicant has analyzed the most severe overpressure transients. The analysis was performed assuming that: (a) the plant is in operation at design conditions of[27] * %[28] of rated steam flow and a reactor vessel dome pressure of * kPag (psig,[29], and (b) the reactor is shut down by ______. The calculated peak pressure at the bottom of the vessel is ______ kPag (______ psig)[30], a value within the code allowable of ______ kPag (______ psig,[31] (110% of vessel design pressure).

2. PWRs

The overpressure protection system prevents overpressurization of the reactor coolant pressure boundary under the most severe transients and limits the reactor pressure during normal operational transients. Overpressurization protection is provided by ______ safety valves. These valves discharge to the pressurizer quench tank through a common header from the pressurizer. The safety and relief valves in the primary, in conjunction with the steam generator safety and relief valves in the secondary, and the reactor protection system, will protect the primary system against overpressure in the event of a complete loss of heat sink.

The peak primary system pressure following the worst transient is limited to the ASME Code allowable (110% of the design pressure) with no credit taken for nonsafety-grade relief systems. The ______ plant was assumed to be operating at design conditions (% of rated power) and the reactor is shut down by a ______ scram. The calculated pressure at the bottom of the vessel is ______ kPag (___ psig)[32], a value within the code allowable of______ kPag (___ psig,[33] (110% of vessel design pressure).

Overpressure protection during low temperature operation (defined in Branch Technical Position RSB 5-2) of the plant is provided by________ .

The applicant has met GDC 15 and 31 and Appendix G ~~since they have~~by implement~~ed~~ing the guidelines of BTP RSB 5-2. In addition, the applicant has incorporated into the~~ir~~[34] design the recommendations of Task Action Plan items II.D.1 and II.D.3~~ of NUREG-0718 and~~ as described in NUREG-0737 (and has met the related requirements of 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi), if applicable).[35]

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*Normally, BWRs are analyzed at 105% rated steam flow at a pressure of 7171 kPag (1040 psig). Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

For design certification reviews, the findings will also summarize, to the extent that the review is not discussed in other safety evaluation report sections, the staff's evaluation of inspections, tests, analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP section.[36]

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR 50 or 10 CFR 52.[37] Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.[38]

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section.[39]

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced NUREGs and Generic Letter.[40]

VI. REFERENCES

1. 10 CFR Part 50, 50.34(f), "Additional TMI-related requirements"[41]

~~1~~2.[42] 10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactor Coolant System Design."

~~8~~3. 10 CFR Part 50, Appendix A, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

~~6~~4. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements."

~~3~~5. Branch Technical Position RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures," attached to this SRP section.

~~4. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License."~~

~~5~~6. NUREG-0737, "Clarification of TMI Action Plan Requirements."

7. NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants."

8. NUREG-1431, "Standard Technical Specifications, Westinghouse Plants."

9. NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants."

10. NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4."

11. NUREG-1434, "Standard Technical Specifications, General Electric Plants, BWR/6."

12. NRC Letter to All Pressurized Power Reactor Licensees, "NUREG-0737 Technical Specifications," (Generic Letter 82-16), September 20, 1982.

13. NRC Letter to All Boiling Water Reactor Licensees, "NUREG-0737 Technical Specifications," (Generic Letter 83-02), January 10, 1983.[43]

14. NRC Letter to All Pressurized Water Reactor Licensees and Construction Permit Holders, "Resolution of Generic Issue 70, 'Power-Operated Relief-Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light Water Reactors,' (Generic Letter No. 90-06)," June 25, 1990.[44]

~~2~~15. ASME Boiler and Pressure Vessel Code, Section III, Article ~~NM-7000, "Protection Against Overpressure,"~~ NB-7000, "Overpressure Protection,"[45] American Society of Mechanical Engineers.

~~7~~16. ASME Boiler and Pressure Vessel Code, Section III, Article ~~NB-7611, "Spring-Loaded Safety Valves."~~ NB-7511.1, "Spring-Loaded Valves."[46]

BRANCH TECHNICAL POSITION RSB 5-2
[Currently the responsibility of the Reactor Systems Branch (SRXB)][47]

OVERPRESSURIZATION PROTECTION OF PRESSURIZED WATER REACTORS WHILE OPERATING AT LOW TEMPERATURES

A. Background

General Design Criterion 15 of Appendix A of 10 CFR Part 50 requires that "the Reactor Coolant System and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."

Anticipated operational occurrences, as defined in Appendix A of 10 CFR Part 50, are "those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power."

Appendix G of 10 CFR Part 50 provides the fracture toughness requirements for reactor pressure vessels under certain conditions. To assure that the Appendix G limits of the reactor coolant pressure boundary are not exceeded during any anticipated operational occurrences, technical specification pressure-temperature limits are provided for operating the plant.
The primary concern of this position is that during startup and shutdown conditions at low temperature, especially in a water-solid condition, the reactor coolant system pressure might exceed the reactor vessel pressure- temperature limitations in the technical specifications established for protection against brittle fracture. This inadvertent overpressurization could be generated by any one of a variety of malfunctions or operator errors. Many incidents have occurred in operating plants as described in Reference 1.

Additional discussion on the background of this position is contained in Reference 1.

B. Branch Position

1. A system should be designed and installed which will prevent exceeding the applicable technical specifications and Appendix G limits for the reactor coolant system while operating at low temperatures. The system should be capable of relieving pressure during all anticipated overpressurization events at a rate sufficient to satisfy the technical specification limits, particularly while the reactor coolant system is in a water-solid condition.

2. The low temperature overpressure protection system should be operable during startup and shutdown conditions below the enable temperature, defined as the water temperature corresponding to a metal temperature of at least $RT_{(NDT)}$ + 50ºC (90ºF)[48] at the beltline location (1/4t or 3/4t) that is controlling in the ~~a~~Appendix[49] G limit calculations.

3. The system should be able to perform its function assuming any single active component failure. Analyses using appropriate calculational techniques must be provided which demonstrate that the system will provide the required pressure relief capacity assuming the most limiting single active failure. The cause for initiation of the event, e.g., operator error, component malfunction should not be considered as the single active failure. The analyses should assume the most limiting allowable operating conditions and systems configuration at the time of the postulated cause of the overpressure event.

   All potential overpressurization events should be considered when establishing the worst-case event. Some events may be prevented by protective interlocks or by locking out power. These events should be identified on an individual basis. If the events are excluded from the analyses, the controls to prevent these events should be in the plant technical specifications.

4. The system should be designed using IEEE Std. ~~-279~~ 603 (Reference 2)[5051] as guidance (see implementation). The system may be manually enabled; however, an alarm to alert the operator to enable the system at the correct plant condition during cooldown~~,~~[52] should be provided. Positive indication should be provided to indicate when the system is enabled. An alarm should be provided when the protective action is initiated. The Instrumentation & Controls Branch (HICB) assists the SRXB in reviews of the design criteria and design for the low temperature overpressure protection system controls and instrumentation as described in subsection I of SRP Section 5.2.2.[53]

5. To assure operational readiness, the overpressure protection system should be testable. Technical specification surveillance requirements should include:

   a. A test performed to assure operability of the system (exclusive of relief valves) prior to each shutdown.

   b. A test for valve operability, as a minimum, be conducted as specified in the ASME Code Section XI.

6. The system must meet the requirements of Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and Section III of the ASME Code.

7. The overpressure protection system should be designed to function during an Operating Basis Earthquake. It should not compromise the design criteria of any other safety-grade system with which it would interface, such that the requirements of Regulatory Guide 1.29, "Seismic Design Classification," are met.

8. The overpressure protection system should not depend on the availability of offsite power to perform its function. The system should be operable from battery-backed power sources, not necessarily Class 1E buses.

9. Overpressure protection systems which take credit for an active component(s) to mitigate the consequences of an overpressurization event should include additional analyses

considering inadvertent system initiation/actuation or provide justification to show that existing analyses bound such an event.

10. If pressure relief is from a low pressure system~~,~~[54] not normally connected to the primary system, the overpressure protection function should not be defeated by interlocks which would isolate the low pressure system from the primary coolant system. (See ~~BTP~~Branch Technical Position ICSB 3 in SRP Chapter 7, Appendix 7-A)[55]

D. REFERENCES

1. NUREG-0138, Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director, NRR, to NRR Staff.

2. IEEE Std. 603-1980, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," (as endorsed by Regulatory Guide 1.153)[56]

SRP Draft Section 5.2.2

Attachment A - Proposed Changes in Order of Occurrence

Item numbers in the following table correspond to superscript numbers in the redline/strikeout copy of the draft SRP section.

Item	Source	Description
1.	Editorial	A punctuation error that caused confusion regarding the meaning of the sentence was corrected. The punctuation is now properly placed between the phrase describing the primary and the phrase describing the secondary.
2.	SRP-UDP format item	Revised Review Interface section of Areas of Review to be consistent with SRP-UDP required format which uses a number/paragraph format to distinguish individual reviews performed by other PRBs.
3.	**Integrated Impact 238.**	Added a review interface with proposed SRP Section 6.8 related to review of PWR depressurization systems. Overpressure protection may be a subsystem or function of the depressurization system.
4.	**PI # 15465,** Editorial	SRP Section 5.2.2 evaluates the over-pressure protection systems that provide protection of the reactor coolant pressure boundary from failure resulting from over-pressure events. There is a direct relationship between the protection against overpressure events and the design of the interfacing systems. Therefore, a review interface with SRP Section 3.12 appears to be appropriate.
5.	Current PRB names and abbreviations.	Editorial change made to reflect current PRB name and responsibility for SRP Section 14.2.
6.	SRP-UDP format item	The review responsibility for SRP Section 3.11 was split out from item 3. in accordance with SRP-UDP guidance.
7.	Editorial	"Review" changed to "reviews" to allow splitting this item into two items.
8.	Editorial	"Installation criteria" was changed to "seismic and dynamic qualification" to properly describe the subject of SRP Section 3.10. This phrase was taken from the title of SRP Section 3.10.
9.	SRP-UDP format item	The review responsibility for SRP Section 3.11 was split out from item 3 in accordance with SRP-UDP guidance. Also, "installation criteria" was changed to "environmental qualification" to properly describe the subject of SRP Section 3.11. This phrase was taken from the title of SRP Section 3.11.

SRP Draft Section 5.2.2

Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
10.	**PI # 19798, PI # 25641**	Added a review interface with SRP Sections 5.2.3, 5.3.1 and 5.3.2. SRP Sections 5.2.3 and 5.3.1 involve the review of the fracture toughness of the RCPB and reactor vessel, respectively. SRP Section 5.3.2 provides for review of the 10 CFR 50, Appendix G, pressure-temperature limits and 10 CFR 50.61 pressurized thermal shock issues. The overpressure protection systems reviewed in SRP Section 5.2.2 are designed to protect the RCPB from overpressure events and maintain the RCPB within the established Appendix G limits.
11.	Current PRB names and abbreviations.	Editorial change made to reflect current PRB abbreviation for the Instrumentation & Controls Branch.
12.	Current PRB names and abbreviations	Editorial change to identify the current PRB names and abbreviation for SRP Section 16.0.
13.	Editorial	Deleted the full title of the branch and referenced only the abbreviation to make this item consistent with the convention utilized in review interface items 2 and 3.
14.	Current PRB names and abbreviations.	Editorial change made to reflect current PRB name and designation for the Quality Assurance and Maintenance Branch.
15.	Editorial	"Boundary" was changed to "it" since boundary is already mentioned as the subject of the phrase. "That" was added before "the probability of..." as the proper article for the phrase. Both changes are to provide clarity and detail to the sentence.
16.	**Integrated Impact 1046**	Restructured and revised the existing paragraph describing TMI Action Item II.D.1 to expand the description of the item, and to incorporate the corresponding requirement from 10 CFR 50.34(f).
17.	**Integrated Impact 1021**	Restructured and revised the existing paragraph describing TMI Action Item II.D.3 to expand the description of the item, and to incorporate the corresponding requirement from 10 CFR 50.34(f).
18.	SRP-UDP format item	Format change to make the citation of references consistent with the SRP-UDP format guidance.
19.	Editorial	A typographical error was corrected by deleting a stray paragraph number that led to confusing interpretation of the subsequent paragraph. It appears that this paragraph should be un-indented and un-numbered so that it applies to the entire preceding paragraph on safety valves.
20.	SRP-UDP format item	Format change to make the citation of references consistent with the SRP-UDP format guidance.

SRP Draft Section 5.2.2

Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
21.	SRP-UDP format item	Format change to make the citation of references consistent with the SRP-UDP format guidance.
22.	SRP-UDP format item	Technical Rationale were developed and added for the following Acceptance Criteria: GDCs 15 and 31. The SRP-UDP program requires that Technical Rationale be developed for the Acceptance Criteria.
23.	Integrated Impact 238	Added new text to incorporate requirements of Generic Letter 90-06 and cite it as a reference.
24.	**Integrated Impact 1099**	Added paragraph to the Review Procedures directing the reviewer to verify compliance with the reporting requirements of TMI Action Item II.K.3.3, and related technical specification guidance.
25.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
26.	NRC Metrication Policy	Reformatted in SI units to be consistent with NRC Metrication Policy.
27.	Editorial	Added "of" to the phrase "design conditions --- * % of rated steam flow.." to provide a proper prepositional phrase. This clarifies the meaning of the sentence.
28.	NRC Metrication Policy	Revised footnote (*) to add the SI equivalent of 1040 psig and reformatted in SI units to be consistent with NRC Metrication Policy.
29.	NRC Metrication Policy	Reformatted in SI units to be consistent with NRC Metrication Policy.
30.	NRC Metrication Policy	Reformatted in SI units to be consistent with NRC Metrication Policy.
31.	NRC Metrication Policy	Reformatted in SI units to be consistent with NRC Metrication Policy.
32.	NRC Metrication Policy	Reformatted in SI units to be consistent with the NRC Metrication policy.
33.	NRC Metrication Policy	Reformatted in SI units to be consistent with NRC Metrication Policy.
34.	Editorial	Grammatical correction and sentence clarification.
35.	**Integrated Impacts 1021 and 1046**	Revised the Evaluation Findings to be consistent with changes to the Acceptance Criteria related to TMI Action Items II.D.1 and II.D.3.
36.	10 CFR 52 applicability related change	Standard design certification (DC) terminology was added to the Evaluation Findings section.

SRP Draft Section 5.2.2

Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
37.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard sentence to address application of the SRP section to reviews of applications filed under 10 CFR Part 52, as well as Part 50.
38.	Editorial	Added paragraph typical of other SRP section IMPLEMENTATION subsections.
39.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
40.	SRP-UDP format item	Added allusion to the fact that Generic Letter contains implementation schedule information.
41.	**Integrated Impact 1021 and 1046**	Added reference to 10 CFR 50.34(f).
42.	SRP-UDP Format Item, Reformat References	Reordered and renumbered references in accordance with SRP-UDP guidance and to accommodate the addition of new references.
43.	**Integrated Impact 1099**	Added reference to the Standard Technical Specification NUREGs, and Generic Letters 82-16 and 83-02. These references were incorporated in the Review Procedures as providing guidance relative to TMI Action Item II.K.3.3.
44.	Integrated Impact 238	Added reference to Generic Letter 90-06.
45.	Editorial	Corrected title of cited reference. There is no Article NM-7000 in section III of the ASME code. The Article on Overpressure Protection has been NB- 7000 since 1973.
46.	Integrated Impact 382	Updated reference to ASME BPVC Section III Article NB-7611.
47.	SRP-UDP Format Item	Added parenthetical identification of the responsible PRB for the Branch Technical Position in accordance with SRP-UDP guidance.
48.	NRC Metrication Policy	Added the SI equivalent to 90ºF and reformatted to be consistent with the NRC Metrication policy.
49.	Editorial	Capitalized "Appendix".
50.	**Integrated Impact 978**	Revise standard citation to reflect the latest version endorsed by the NRC.
51.	Integrated Impact 555 SRP-UDP standards citation update	Consideration should be given to updating the citation of IEEE 279 pending the review and approval of the associated standard comparison.
52.	Editorial	Deleted a comma to clarify the meaning of the sentence.
53.	Editorial	Added description of the HICB review described in Review Interface 6, subsection I, of SRP Section 5.2.2.

Item	Source	Description
54.	Editorial	Deleted a comma to clarify the meaning of the sentence.
55.	SRP-UDP format item, reference verification	Spelled out "branch technical position" and identified its location in the SRP to improve this reference citation.
56.	**Integrated Impact 978**	Added reference to IEEE Std. 603-1980. This standard replaces the existing citation of IEEE Std. 279 in B.4 of the BTP.

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SRP Draft Section 5.2.2

Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
238	Recommend revision to the SRP to incorporate Generic Letter 90-06. The staff used Generic Letter 90-06 to resolve GSIs 70 and 94 which relate to reliability of PORVs and PORV block valves, and availability of low temperature overpressure protection systems (LTOP).	SRP 5.2.2, Sections III.6, III.7, and VI.9.
382	Recommend revision of the SRP reference section to cite the number/title of the accepted edition of Section III of the ASME Code. The proposed edition to be referenced was accepted by the NRC in 10CFR50.55a.	SRP 5.2.2, Section VI.7.
555	Upon final approval of the standard comparison, recommend revising the citation of IEEE 279 to reference the most current standard.	This is a placeholder integrated impact and will not be processed further at this time.
978	Update the citation in of IEEE Std 279 in BTP RSB 5-2 to cite IEEE Std 603-1980	BTP RSB 5-2, paragraph B.4 and D.2
1021	Revise Acceptance Criteria related to TMI action plan item II.D.3 regarding direct indication of relief and safety valve position.	II. Acceptance Criteria; IV. Evaluation Findings; and VI. References
1046	Revise Acceptance Criteria related to TMI action plan item II.D.1 regarding qualification of reactor coolant system relief and safety valves .	II. Acceptance Criteria; IV. Evaluation Findings; and VI. References
1099	Revise Acceptance Criteria and Review Procedures to incorporate TMI action plan item II.K.3.3 regarding reporting of safety and relief valve failures and challenges.	Review Procedures; and References
1503	Revise the SRP to address the Generic Letter 83-10 guidance related to RCP automatic trips and associated criteria for PORVs.	Not to be processed further.